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Radiation Dose Assessments for Materials with Elevated Natural Radioactivity

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SUMMARY

This report provides practical information needed for evaluating the radiation dose to the general public and workers caused by materials containing elevated levels of natural radionuclides. The report presents criteria, exposure scenarios and calculations used to assess dose with respect to the safety requirements set for construction materials in accordance with the Finnish Radiation Act.

A method for calculating external gamma exposure from building materials is presented in detail. The results for most typical cases are given as specific dose rates in table form to enable doses to be assessed without computer calculation. A number of such dose assessments is presented, as is the corresponding computer code. Practical investigation levels for the radioactivity of materials are defined.

PREFACE

The radioactivity of building materials, industrial by-products and waste has been investigated by the Finnish Centre for Radiation and Nuclear Safety (STUK) for some 20 years. As a result, STUK now has a detailed view of Finns' exposure to these sources of natural radiation. Most of these investigations have been conducted by Dr. Raimo Mustonen.

Finnish radiation legislation was revised in 1992, taking into account the 1990 ICRP recommendations. Practices causing occupational or public exposure to natural radiation were considered. On the basis of the new legislation, STUK has issued general guides concerning exposure to natural radiation. The radiation protection criteria adopted in the ST Guide 12.2 'Radioactivity of Construction materials, Fuel Peat and Peat Ash' are based on the work of Dr. Mustonen who has also presented the activity indexes for construction materials given in Sections 2.4 and 5.1. All other criteria and exposure scenarios presented in and assessments performed for this report were done by the author.

CONTENTS

1 INTRODUCTION

Exposure to natural sources of radiation is often influenced or can be influenced by human activities. Building materials, for instance, cause excess external gamma exposure due solely to their influenced exposure geometry when compared with that of the undisturbed earth's crust. Such excess in exposure is commonly excluded from any system of radiological protection. Construction material can, however, cause substantial radiation exposure if they contain elevated levels of naturally occurring radionuclides.

This report provides the practical information needed for evaluating public and occupational radiation doses caused by materials containing elevated levels of natural radionuclides. The main objective is to present the criteria and calculations used in assessing doses with respect to the safety requirements set for construction materials in accordance with the Finnish Radiation Act. Investigation levels for materials causing excess exposure are also derived.

A method for calculating the external gamma dose from building materials is presented, and the results for most typical cases are given as specific dose rates in tabular form. This allows the most typical doses to be assessed without computer calculation. Several examples of such assessments are given in Appendix 2. The calculation method is described in detail and the corresponding computer code used is presented in Appendix 1.

The approach selected for dose assessment here is rather theoretical and is considered best for supervisory purposes. When an authority has to decide on the acceptability of a material for certain use, the activity concentrations are usually the only measured parameters available. Dose assessment must therefore be performed on the basis of different theoretical exposure scenarios.

If, however, the data measured or otherwise known on external dose rate, dust and radon concentrations, soil-to-plant transfer factors, food consumption profiles etc. for a critical group are available, these are preferred. Theoretical dose assessments provide a perspective on the relative importance of different exposure pathways and can show where more elaborate assessments, including site-specific measurements etc., are needed.

2 FINNISH LEGISLATION ON NATURAL RADIATION

2.1 Scope of the Radiation Act

Finnish radiation legislation was revised in 1992, taking into account the 1990 ICRP recommendations[1]. Practices causing occupational or public exposure to natural radiation were considered. The Radiation Act (592/91) states that radiation practices also comprise operations or circumstances in which a person's exposure to natural radiation causes or may cause a health hazard.

Under the Radiation Act radiation practices comprise the production, trade in or handling of materials with elevated natural radioactivity causing significant excess exposure to the general public or workers. Radon in the workplace and other significant occupational exposure to natural radiation sources and public exposure to sources such as household water with elevated natural radioactivity supplied by water works, fall within the scope of the Act.

Radon in homes does not come within the scope of the Act. The Ministry of Social Affairs and Health has issued separate provisions on the upper limits for radon concentrations in homes (944/92). Accordingly, the upper limit for existing buildings is 400 Bq m^{-3} and the design level for new buildings 200 Bq m^{-3}.

According to the Radiation Act, when utilizing natural resources that contain radioactive materi-als, the responsible party shall ensure that the radioactive wastes do not pose any health or environmental hazards during operations, including its final stages.

2.2 Radiation safety guides for natural radiation

Annual dose limits based on the 1990 ICRP recommendations[1] were added to the Radiation Decree (1512/91). The Radiation Act authorizes STUK to issue lower dose constraints in individual cases to keep the radiation exposure as low as reasonably achievable. STUK is also permitted to provide general guidance on how to attain the level of safety prescribed by the Act.

On this basis, STUK has issued general guides (ST Guides) setting out safety requirements for different practices causing excess exposure to natural radiation. The safety requirements, expressed as maximum annual doses, are based on assessments at the national level leading to only a small number of affected enterprises.

Radon in workplaces is the most significant source of occupational exposure to natural sources of radiation in Finland. The upper limit for radon concentration in workplaces is issued in the ST-guide 1.2 'Application of Maximum Radiation Exposure Values and Monitoring of Radiation Exposure.' The limit, 400 Bq m^{-3}, is the annual mean concentration during working hours in a job with regular working hours. If less than 600 hours are worked a year, a higher radon concentration level may be accepted.

If the radon concentration exceeds 400 Bq m^{-3} in a workplace the employer must take action to lower it. If, despite of sufficient efforts,it cannot be reduced below 400 Bq m^{-3}, the work is classified as radiation work. In radiation work radon exposures must be recorded and heath surveillance arranged for the workers. The maximum radon concentration in radiation work is 3 200 Bq m^{-3}.

ST Guide 12.1 'Radiation Safety in Mining and Underground Excavation' defines the upper limits for radon exposure in mines and underground excavations. The limits are the same as those in other workplaces, except that regular monitoring in these workplaces is required.

The radiation safety requirements and corresponding investigation levels for household water are prescribed in the ST Guide 12.3 'Radioactivity of Household Water'. The safety requirement is set at 0.5 mSv (annual effective dose to the consumer). The guide applies to water works and to professional producers of beverages and foods relying on their own water supply. It does not cover the well or equipment of an individual.

The guide for construction materials is described in greater detail in Sections 2.3 and 2.4. More information on and findings from applying the guidance for natural radiation are presented in reference articles.[2-4]

2.3 Safety requirements for construction materials

ST Guide 12.2 'Radioactivity of Building Materials, Fuel Peat and Peat Ash' presents the safety requirements for radiation exposure for building materials and materials used in road and related construction. It also covers production of fuel peat and the handling and disposal of peat ash. The safety requirements are given as the annual effective dose:

Cause or target of exposure	**Effective dose***
a) Exposure to workers	
Handling of peat or peat ash	1 mSv
b) Exposure to public	
Building materials (radon exposure excluded)	1 mSv
Use of peat ash as additive in building materials	0.1 mSv
Materials used for constructing streets, playgrounds and landfills and for landscaping, etc	0.1 mSv

*The excess exposure due to natural radiation. Mean outdoor exposure originating from radionuclides in the undisturbed earth's crust is subtracted in all dose assessments.

The purpose of issuing safety requirements is to limit the radiation exposure caused by the use of materials containing elevated levels of natural radionuclides. The goal is to eliminate the most extreme cases of public or occupational exposure. The safety requirements are now very specific references for reducing radiation exposure to natural sources in the most extreme individual cases.

2.4 Activity indexes for construction materials

Activity indexes are used to assess whether the safety requirements are being fulfilled. For construction materials the activity indexes are calculated on the basis of the measured activity concentrations of radium (^{226}Ra), thorium (^{232}Th) and potassium (^{40}K). In special cases other nuclides are also considered, e.g. caesium (^{137}Cs) in fuel peat ash. The exposure scenarios and criteria used in their derivation are given in Section 5.1. The activity indexes are presented in Table *I*.

If the value of the activity index is 1 or less, the corresponding material can be used, with regard to radioactivity, without restriction. If the value exceeds 1, the responsible party (producer or dealer) is required to assess the radiation exposure caused by the material and show specifically that the safety requirement is fulfilled. If necessary STUK shall issue instructions on limiting the exposure to radiation. If the responsible party fails to carry out such an investigation, STUK is authorized to issue an order to this effect.

The guidance for construction materials does not include the radon release to indoor air from building materials. The soil under a building is usually the major source of indoor radon. Limits to radon concentrations have been set separately for workplaces and homes. They are applied without regard to source, soil or building material, and therefore separate limitations have not been issued for radon release from building materials.

The activity index I_1 for building materials omits the extreme cases of radon release as it effectively restricts the ^{226}Ra concentration to a level of about 150 Bq kg^{-1} (as some ^{232}Th and ^{40}K always occur in the material, a level of 300 Bq kg^{-1} of ^{226}Ra is most unlikely).

Table *I*. Activity indexes for construction materials. C_{Th}, C_{Ra}, C_K and C_{Cs} are the activity concentrations of the material, expressed in Bq kg^{-1}.

Building materials	$I_1 = \frac{C_{Th}}{200\ Bq/kg} + \frac{C_{Ra}}{300\ Bq/kg} + \frac{C_K}{3000\ Bq/kg}\ 1$
Materials used for streets and playgrounds	$I_2 = \frac{C_{Th}}{500\ Bq/kg} + \frac{C_{Ra}}{700\ Bq/kg} + \frac{C_K}{8000\ Bq/kg} + \frac{C_{Cs}}{2000\ Bq/kg}\ 2$
Materials used for land filling	$I_3 = \frac{C_{Th}}{1500\ Bq/kg} + \frac{C_{Ra}}{2000\ Bq/kg} + \frac{C_K}{20000\ Bq/kg} + \frac{C_{Cs}}{5000\ Bq/kg}\ 3$
Handling of peat ash	$I_4 = \frac{C_{Th}}{3000\ Bq/kg} + \frac{C_{Ra}}{4000\ Bq/kg} + \frac{C_K}{50000\ Bq/kg} + \frac{C_{Cs}}{10000\ Bq/kg}\ 4$

3 METHODOLOGY FOR DOSE ASSESSMENT

3.1 Exposure to external gamma radiation

The method used for calculating the external gamma dose rate is similar to that presented by Stranden[5] and Mustonen[6]. The dose rate is calculated for a rectangular source of uniform density and activity concentration. The indoor dose rate is calculated by summing the separately calculated dose rates caused by walls, floor and ceiling.

The calculation method presented in Appendix 1 is elaborated to cover situations in which the source may comprise two layers of material with different densities and activity concentrations. This allows the source origin of gamma dose to be estimated, for instance, if concrete walls are covered with a thin layer of other material such as tiles, etc.

The calculation model was tested against some published results. Specific absorbed dose rates in air were calculated for a room in a concrete building. The specific absorbed dose rates outdoors due to radioactivity in the soil were also calculated. The results of the comparisons are given in Tables *II* and *III*.

A conversion factor of 0.7 Sv Gy^{-1} given in the UNSCEAR 1993 report[7] is used for converting the absorbed dose in air to the effective dose. The same value is used for all gamma emitters discussed in this report (natural radionuclides and ^{137}Cs).

3.2 Inhalation and ingestion dose

The committed effective dose due to inhalation of radionuclides is calculated by using the formula:

$$E_{inh} = e_{inh}\, A_d\, c_d\, v\, t \qquad (1)$$

where the parameters are:

E_{inh} committed effective dose, Sv
e_{inh} inhalation dose coefficient, Sv · Bq^{-1}
v breathing rate, m^3 h^{-1}
A_d activity concentration of the dust, Bq g^{-1}
c_d dust concentration in air, g · m^{-3}
t duration of exposure, h

The committed effective dose due to ingestion of radionuclides is calculated by using the formula:

$$E_{ign} = e_{ign}\, A_p\, m \qquad (2)$$

where the parameters are:

E_{ign} committed effective dose due to ingestion, Sv
e_{ign} ingestion dose coefficient, Sv · Bq^{-1}
m amount of the product consumed, g
A_p activity concentration of the product, Bq g^{-1}

The inhalation and ingestion dose coefficients based on the latest ICRP annals[11] (Publication 68) for natural radionuclides are given in Table *IV*. Natural radionuclides usually occur in secular equilibrium with their daughter nuclides. Dose coefficients for typical nuclide compositions are given in Table *V*.

3.3 Radon exposure

By definition, the ratio of potential alpha energy exposure to the equilibrium equivalent exposure is $5.56 \cdot 10^{-9}$ J h m^{-3} per Bq h m^{-3} (ICRP 65[12]). If the equilibrium factor is 0.5, the ratio of potential alpha energy exposure to radon exposure is $2.78 \cdot 10^{-9}$ mJ h m^{-3} per Bq h m^{-3}.

In the ICRP 65[12] it has been estimated that a potential alpha energy exposure of 1 mJ h m^{-3} causes an effective dose of 1.1 mSv at home and 1.4 mSv at work. On this basis, radon exposure of 1 Bq h m^{-3} causes an effective dose of $3.1 \cdot 10^{-9}$ Sv at home and $3.9 \cdot 10^{-9}$ Sv at work. The conversions of Table *VI* are obtained by using occupancy factors of 1 600 h a^{-1} at work and 7 000 h a^{-1} at home .

All materials containing ^{226}Ra release some radon into the surrounding air. Building materials therefore cause an excess in indoor air radon concentration, which can be described with the equations:

$$C = \frac{G}{nV} \; ; \quad G = F \cdot E = \lambda\pi A \cdot M \qquad (3)$$

where the parameters are:

- C excess of indoor radon concentration, Bq m^{-3}
- G rate of radon entry into the room from building materials, Bq h^{-1}
- n ventilation rate, h^{-1}
- V volume of the room, m^{-3}
- F total exhaling area, m^{-2}
- E area exhalation rate of the material, Bq h^{-1} m^{-2} (see Table *VIII* for typical values)
- λ physical decay constant of radon, 0.00756 h^{-1}
- π radon emanation coefficient (see Table *VII* for typical values)
- A ^{226}Ra concentration of the material, Bq kg^{-1}
- M total mass of the material, kg

The rate of radon entry into the room, G, can be calculated on the basis of the exhaling area (F A) or the radium concentration and total mass of the material ($\lambda\pi A$ M). The latter method of calculating leads to an upper limit for the entry rate, because it does not consider whether all radon released to the pore space within the material will be released from the surface, since some will decay before it reaches the surface (limited diffusion length).

Typical values for the factors used in estimating the indoor air radon concentration from building materials are given in Tables *VII* and *VIII*. The design value for the ventilation rate in new buildings is often 0.5 h^{-1}. In most buildings the ventilation rate is in the range 0.2…1 h^{-1}.

Table II. *Absorbed dose rate in indoor air for decay series of ^{238}U and ^{232}Th and the nuclide ^{40}K. Room dimensions 5 m x 4m x 2.8 m. Walls, floor and ceiling are 20-cm-thick concrete, with a density of 2 320 kg m^{-3}.*

Reference	**Specific absorbed dose rate nGy h^{-1} per Bq kg^{-1}**			**Method**
	^{238}U	**^{232}Th**	**^{40}K**	
Koblinger[8] (1978)	0.922	1.02	0.0779	Monte Carlo
Stranden[5] (1979)	0.914	1.10	0.0776	Attenuation and build up
Mustonen[6] (1985)	0.922	1.10	0.0806	Attenuation and build up
This work	0.908	1.06	0.0767	Attenuation and build up

Table III. *Absorbed dose rate in outdoor air for decay series of ^{238}U and ^{232}Th and the nuclides ^{40}K and ^{137}Cs uniformly distributed in the soil.*

Reference	**Specific absorbed dose rate in air nGy h^{-1} per Bq kg^{-1}**				**Method**
	^{238}U	**^{232}Th**	**^{40}K**	**^{137}Cs**	
Beck[9] (1972)	0.430	0.666	0.0422		Monte Carlo
Saito[10] (1995)	0.463	0.604	0.0417		Monte Carlo
This work	0.470	0.572	0.0421	0.156	Attenuation and build up

Table IV. *Effective dose coefficients for ingested and inhaled particles. Poorly soluble oxides are assumed for inhalation and soluble compounds for ingestion. Data from ICRP 68*[11]*. AMAD is the activity median aerodynamic diameter of the inhaled particles.*

Nuclide	**Inhalation, Sv Bq^{-1}**		**Ingestion, Sv Bq^{-1}**
	AMAD=1 µm	AMAD=5 µm	
^{238}U	$7.3 \cdot 10^{-6}$	$5.7 \cdot 10^{-6}$	$4.4 \cdot 10^{-8}$
^{234}Th	$7.3 \cdot 10^{-9}$	$5.8 \cdot 10^{-9}$	$3.4 \cdot 10^{-9}$
^{234}Pa	$4.0 \cdot 10^{-10}$	$5.8 \cdot 10^{-10}$	$5.1 \cdot 10^{-10}$
^{234}U	$8.5 \cdot 10^{-6}$	$6.8 \cdot 10^{-6}$	$4.9 \cdot 10^{-8}$
^{230}Th	$1.3 \cdot 10^{-5}$	$7.2 \cdot 10^{-6}$	$2.1 \cdot 10^{-7}$
^{226}Ra	$1.6 \cdot 10^{-5}$	$1.2 \cdot 10^{-5}$	$2.8 \cdot 10^{-7}$
^{210}Pb	$8.9 \cdot 10^{-7}$	$1.1 \cdot 10^{-6}$	$6.8 \cdot 10^{-7}$
^{210}Bi	$8.4 \cdot 10^{-8}$	$6.0 \cdot 10^{-8}$	$1.3 \cdot 10^{-9}$
^{210}Po	$3.0 \cdot 10^{-6}$	$2.2 \cdot 10^{-6}$	$2.4 \cdot 10^{-7}$
^{232}Th	$2.3 \cdot 10^{-5}$	$1.2 \cdot 10^{-5}$	$2.2 \cdot 10^{-7}$
^{228}Ra	$2.6 \cdot 10^{-6}$	$1.7 \cdot 10^{-6}$	$6.7 \cdot 10^{-7}$
^{228}Ac	$1.4 \cdot 10^{-8}$	$1.2 \cdot 10^{-8}$	$4.3 \cdot 10^{-10}$
^{228}Th	$3.9 \cdot 10^{-5}$	$3.2 \cdot 10^{-5}$	$7.0 \cdot 10^{-8}$
^{224}Ra	$2.9 \cdot 10^{-6}$	$2.4 \cdot 10^{-6}$	$6.5 \cdot 10^{-8}$
^{235}U	$7.7 \cdot 10^{-5}$	$6.1 \cdot 10^{-6}$	$4.6 \cdot 10^{-8}$
^{231}Th	$3.2 \cdot 10^{-10}$	$4.0 \cdot 10^{-10}$	$3.4 \cdot 10^{-10}$
^{231}Pa	$3.2 \cdot 10^{-5}$	$1.7 \cdot 10^{-5}$	$7.1 \cdot 10^{-7}$
^{227}Ac	$6.6 \cdot 10^{-5}$	$4.7 \cdot 10^{-5}$	$1.1 \cdot 10^{-6}$
^{227}Th	$9.6 \cdot 10^{-6}$	$7.6 \cdot 10^{-6}$	$8.9 \cdot 10^{-9}$
^{223}Ra	$6.9 \cdot 10^{-6}$	$6.9 \cdot 10^{-6}$	$1.0 \cdot 10^{-7}$

Table V. *Inhalation and ingestion dose coefficients for different nuclide combinations, in μSv Bq^{-1}. Calculated from data given in Table IV.*

Nuclide or combination	**Inhalation, μSv Bq^{-1} AMAD 1 μm**	**5 μm**	**Ingestion, μSv Bq^{-1}**	**Remarks**
^{238}U	7.3	5.7	0.044	Without decay products
^{238}U+	29	20	0.31	Nuclides ^{238}U,^{234}Th,^{234m}Pa,^{234}U and ^{230}Th included
^{238}U++	49	35	1.5	Nuclides ^{238}U+ and ^{226}Ra+ included.
U_{nat}	55	39	1.6	Nuclides ^{238}U++ and ^{235}U+ included. Activity ratio ^{238}U:^{235}U = 20:1 assumed. Factors refer to activity of ^{238}U.
^{226}Ra	16	12	0.28	Without decay products
^{226}Ra+	20	15	1.2	Nuclides ^{226}Ra and ^{210}Pb+ included. Nuclides ^{222}Rn,^{218}Po,^{214}Pb,^{214}Bi and ^{214}Po are not included.
^{210}Pb	0.89	1.1	0.68	Without decay products
^{210}Pb+	4.0	3.4	0.92	Nuclides ^{210}Pb,^{210}Bi and ^{210}Po included.
^{210}Po	3.0	2.2	0.24	
^{232}Th	23	12	0.22	Without decay products
^{232}Th+	68	48	1.0	Nuclides ^{232}Th,^{228}Ra,^{228}Ac,^{228}Th and ^{224}Ra included
^{235}U	7.7	6.1	0.046	Without decay products
^{235}U+	120	85	2.0	Nuclides ^{235}U,^{231}Th,^{231}Pa,^{227}Ac, ^{227}Th and ^{223}Ra included

Table VI. *Dose conversions used for radon exposure and parameters used in their derivation.*

	At work	**At home**
Occupancy factor	1 600 h	7 000 h
Equilibrium factor	0.5	0.5
Effective dose per potential alpha energy exposure	1.4 mSv per mJ h m^{-3}	1.1 mSv per mJ h m^{-3}
Effective dose per radon exposure	$4 \cdot 10^{-9}$ Sv per Bq h m^{-3}	$3 \cdot 10^{-9}$ Sv per Bq h m^{-3}
Effective dose per radon concentration	6 µSv per Bq m^{-3}	20 µSv per Bq m^{-3}

Table VII. *Radon emanation coefficients (π) for some building materials*

Material		**Emanation coefficient, dimensionless**		**References**
		Typical value	**Variations**	
Concrete		0.20	0.05…0.3	13, 14, 15, 16, 17
Red brick		0.02	0.01…0.05	13, 14, 15, 16, 17
By-product gypsum		0.01	0.01…0.3	14, 17
Lightweight expanded clay aggregate		0.2	0.1…0.4	15, 17
Soil	Clay	0.3	0.2…0.5	18
	Sand	0.2	0.1…0.4	18
	Gravel	0.1	0.05…0.3	18
	Till	0.2	0.1…0.4	18

Table VIII. *Specific area exhalation rate for a 20-cm-thick slab of material (exhalation from one side only).*

Material	**Specific exhalation rate, Bq $h^{-1}m^{-2}$ per Bq kg^{-1}**		**References**
	Typical value	**Variations**	
Concrete	0.4	0.1…0.5	13, 14, 15, 16, 17
Red brick	0.03	0.01…0.07	13, 14, 15, 16, 17
Brick	0.07	0.03…0.15	15, 16, 17
By-product gypsum	0.1	0.05…0.1	16, 17
Lightweight expanded clay aggregate	0.1	0.05…0.1	15, 17

4 ESTIMATION OF RADIATION DOSE

4.1 General approach

When limits are set for exposure to natural radiation sources it must be clearly defined as to what extent the exposure caused by 'normal' background is included. The safety requirements for construction materials are defined as the excess exposure caused by these materials. In all dose assessments, then, the exposure from natural radionuclides in the undisturbed Earth's crust and cosmic radiation are subtracted.

The basic concept in determining the excess exposure is the following. The total exposure caused by the material and the influenced background is first evaluated. The exposure caused by the background before any human influence is then subtracted from it. This result is referred to as the excess exposure.

In individual cases such as mining disposal areas etc., exposure caused by uninfluenced background can be evaluated on site before the activities or later by monitoring the surroundings of the site. In the case of building materials, the place of use is not known beforehand and therefore a more general subtraction of background on the basis of national or areal averages must be performed.

The population-weighted mean terrestrial dose rate outdoors in Finland is 71 nGy h^{-1} [19]. A rounded value of 70 nGy h^{-1} is subtracted from the calculated dose rates. In some parts of Finland[20], the external gamma dose due to the ^{137}Cs fallout from the Chernobyl accident should be treated as 'existing background' which should be subtracted from the assessed dose rates. The possible shielding effect of materials for cosmic radiation is considered small, and therefore exposure originating from cosmic radiation is excluded in all assessments.

The safety requirement for landfill and landscaping presented in ST Guide 12.2 considers only external gamma radiation. At large mining tailings and other similar areas the discharges of radionuclides to the environment and all radiation exposure pathways shall also be considered.

4.2 Building materials

The gamma dose rate is calculated in the middle of a room presented in Figure 1. Mustonen[6] analyzed the isodose rate curves in a typical room and showed that the variation in dose rate is small (5 - 10%) and the dose rate in the middle of the room is a good approximation for the average dose rate in the room. The specific dose rates for walls, floor and ceiling are given in Table *IX*. The indoor dose rate is calculated by summing the separately calculated dose rates caused by walls, floor and ceiling. Dose assessments resulting from the use of the specific dose rates of Table *IX* are described in Examples 1 and 2 of Appendix 2. An assessment of excess radon exposure due to building materials is given in Example 3 of Appendix 2.

4.3 Landfill and disposal areas

The specific gamma dose rates outdoors are given in Table *X*. In the case of a covering layer on top of the materials concerned the outdoor dose rate is calculated by summing the separately calculated dose rates caused by the cover and the material concerned. Dose assessment by using the specific dose rates of Table *X* is described in Example 4 of Appendix 2.

At large disposal areas, exposure through the food chain and from inhalation of dust must be considered. The radiation exposure should be evaluated for a critical group of people by using estimated consumption profiles and estimations on the possible production of goods in the area. Such assessments are very difficult to perform, however, because of the lack of information on future use of the area (hundreds or thousands of years) and site-specific values for soil-to-plant transfer factors.

A very rough estimate for possible ingestion dose can be obtained in the following manner. The average intake of natural radionuclides to the body through the diet has been estimated in the UNSCEAR 1993 report[4] by using an average food consumption profile and average activity concentrations in dietary materials. By using the effective dose coefficients given in ICRP 68[8] we can calculate that the uranium and thorium series radionuclides cause average annual effective doses 41 μSv and 9 μSv, respectively.

These doses are ultimately proportional to the average activity concentrations of ^{238}U and ^{232}Th in soil. The global average concentrations of ^{238}U and ^{232}Th in soil are about 40 Bq kg^{-1}. On this basis it can be postulated that 1 Bq kg^{-1} of ^{238}U and ^{232}Th in soil ultimately cause annul effective doses of 1 μSv and 0.2 μSv, respectively.

Of course this is a very rough estimate, and it should be realized that any such generalization with averaged values will very likely lead to faulty conclusions in individual cases. If no other reference is available, however, such a reference conversion can give a view of the possible order of magnitude of the ingestion dose.

In the above estimations it is assumed that all the goods consumed by a critical group would be produced in the area, this is unlikely however. It could be postulated that e.g. 1 - 10 % of the goods consumed by a critical group are produced there, depending on the size of the area. This should be considered when applying the conversions given above.
If the active material is not covered with any other material the possible inhalation dose must also be considered. The dose is calculated by using Equation (1). If no measured data are available the parameter values presented in Table *XII*I can be used.

4.4 Operations with and handling of materials

In operations with and in handling of materials the most likely exposure pathways are external gamma exposure and inhalation of dust. Exposure caused by a semi-infinite source is in most cases an upper limit for gamma exposure. This can be calculated using the specific absorbed dose rates given in Table *X*.

A more typical situation occur when the exposure is caused by a source more restricted in geometry, e.g. as a pile of material. The gamma exposure caused by a pile of material depends on the amount of material and the distance from it. The gamma rays causing the exposure usually originate only in the top layers facing the exposed person. As a gamma ray source, therefore, a material pile is better described by its estimated facing area than by its mass.

The specific absorbed dose rates in air from a pile of material with different facing areas and for different distances between the pile and the place of exposure are given in Table *XI*. Dose assessment obtained by use of these data is described in Example 5 of Appendix 2. If the material can raise dust, the possible inhalation dose is calculated by using Equation (1). If no measured data are available the parameter values presented in Table *XIII* can be used. Radon exhalation and radon exposure excess are calculated by using Equation (3), as been presented in the Example 3 of Appendix 2.

Table IX. *Specific dose rate in air from the different structures in the room of Figure 1.*

Specific mass of wall material kg m^{-2}	**Wall material (top layer)** pGy h^{-1} per Bq kg^{-1} ^{226}Ra	^{232}Th	^{40}K	^{137}Cs	**20 cm thick concrete behind the wall material** pGy h^{-1} per Bq kg^{-1} ^{226}Ra	^{232}Th	^{40}K	^{137}Cs
Wall W1 Dimensions 12.0 m x 2.8 m, distance 3.5 m								
0	0	0	0	0	95	110	8.0	32
25	9	10	0.73	3.1	87	100	7.3	30
50	18	21	1.5	6.2	80	94	6.7	27
100	35	40	2.8	12	65	77	5.6	22
150	50	56	3.9	17	52	62	4.6	17
200	61	70	4.9	21	40	50	3.8	13
300	79	91	6.4	27	24	31	2.5	7.6
500	96	110	8.1	32	8	12	1.0	2.3
Wall W2 Dimensions 7.0 m x 2.8 m, distance 6.0 m								
0	0	0	0	0	32	37	2.7	11
25	2.7	3.1	0.22	0.93	30	35	2.5	10
50	5.5	6.2	0.44	1.9	28	32	2.3	9.4
100	11	12	0.85	3.7	22	27	2.0	8.0
150	15	18	1.2	5.3	19	23	1.7	6.5
200	20	22	1.6	6.7	16	19	1.4	5.2
300	26	30	2.1	8.8	10	13	1.0	3.2
500	33	38	2.7	11	3.7	5.4	0.45	1.1
Floor or Ceiling Dimensions 12.0 m x 7.0 m, distance 1.4 m								
0	0	0	0	0	350	410	30	120
25	46	52	3.7	16	310	370	27	110
50	90	100	7.1	31	270	330	24	92
100	160	190	13	56	200	250	18	67
150	220	250	18	75	150	180	14	48
200	260	300	21	89	110	140	11	34
300	310	360	26	105	56	78	6.3	17
500	350	420	30	120	15	27	2.2	4.4

Table X. *Specific absorbed dose rate outdoors.*

Specific mass kg m^{-2}	Material layer on top pGy h^{-1} per Bq kg^{-1}				Soil under the layer pGy h^{-1} per Bq kg^{-1}			
	^{226}Ra	^{232}Th	^{40}K	^{137}Cs	^{226}Ra	^{232}Th	^{40}K	^{137}Cs
Soil Dimensions 20 m x 20 m, distance 1.0 m								
0	0	0	0	0	470	570	42	160
25	83	95	6.6	29	400	490	36	130
50	150	170	12	52	330	410	30	110
100	250	290	20	85	230	300	22	77
200	360	420	30	120	120	160	13	38
300	410	480	35	140	62	91	7.3	19
400	440	520	38	150	33	54	4.3	9.6
500	450	540	40	150	17	32	2.6	4.9

Table XI. *Specific absorbed dose rate in air from a pile of material*

Facing area m^2	^{226}Ra	^{232}Th	^{40}K	^{137}Cs	^{226}Ra	^{232}Th	^{40}K	^{137}Cs
	pGy h^{-1} per Bq kg^{-1}				pGy h^{-1} per Bq kg^{-1}			
	Distance 1 m				Distance 2 m			
1	56	66	4.8	19	17	21	1.5	5.8
4	160	190	14	52	60	72	5.3	20
25	330	400	30	110	210	250	18	69
∞	470	570	42	160	470	570	42	160
	Distance 5 m				Distance 10 m			
1	3.1	3.7	0.27	1.0	0.79	1.0	0.07	0.26
4	12	14	1.1	4	3.1	3.8	0.28	1.1
25	64	76	5.6	21	19	22	1.7	6.3
∞	470	570	42	160	470	570	42	160

5 DERIVATION OF INVESTIGATION LEVELS

When materials containing elevated levels of natural radionuclides are utilized the possible radiation dose caused must be assessed with respect to the safety requirements described in Title 2. For this purpose it is useful to derive some easily monitored triggers indicating the need for further assessments. These are called investigation levels, and are expressed in measurable quantities such as activity concentrations or dose rates. Activity concentrations can be presented as activity indexes which consider all the different radionuclides in the material.

5.1 Activity indexes for construction materials

The activity indexes given in Section 2.3 have been derived to indicate whether the safety requirements given in Section 2.2 are being fulfilled. The general criteria and parameter values used are presented in Table *XII*. The calculations in deriving the activity indexes are:

Building materials

$(0.92 \cdot C_{Ra}\text{-}70) \cdot 10^{-9}$ Gy h^{-1} $\cdot$ 0.7 Sv Gy^{-1}
$\cdot$ 7 000 h a^{-1} = 10^{-3} Sv a^{-1}
=> C_{Ra} = 270 Bq kg^{-1}

The activity concentrations C_{Th} = 226 Bq kg^{-1} and C_K = 3 069 Bq kg^{-1} are calculated similarly, leading to the activity index I_1 for building materials presented in Section 2.3.

Materials used for constructing streets and playgrounds

$(0.46 \cdot C_{Ra}\text{-}70) \cdot 10^{-9}$ Gy h^{-1} $\cdot$ 0.7 Sv Gy^{-1}
$\cdot$ 500 h a^{-1} = 10^{-4} Sv a^{-1}
=> C_{Ra} = 773 Bq kg^{-1}

The activity concentrations C_{Th} = 574 Bq kg^{-1}, C_K = 8 683 Bq kg^{-1} and C_{Cs} = 1 905 Bq kg^{-1} are calculated similarly, leading to the activity index I_2 for materials used on streets and playgrounds presented in Section 2.3. Note that in the case of ^{137}Cs the background 70 nGy h^{-1} is not subtracted.

Materials used for landfill

$(0.46 \cdot C_{Ra}\text{-}70) \cdot 10^{-9}$ Gy h^{-1} $\cdot$ 0.7 Sv Gy^{-1}
$\cdot$ 150 h a^{-1} = 10^{-4} Sv a^{-1}
=> C_{Ra} = 2 222 Bq kg^{-1}

The activity concentrations C_{Th} = 1 649 Bq kg^{-1}, C_K = 24 900 Bq kg^{-1} and C_{Cs} = 6 347 Bq kg^{-1} are obtained similarly, leading to the activity index I_3 for materials used in landfill presented in Section 2.3. Note that in the case of ^{137}Cs the background 70 nGy h^{-1} is not subtracted.

Handling of peat ash

A power plant burning peat produces thousands of tonnes of peat ash every year. The ash is usually delivered by truck to some disposal area or landfill nearby. The truck drivers are the critical group for radiation exposure. The ash is slightly wet when delivered, so dusting is reduced significantly. The main source of exposure is direct gamma radiation.

The truck drivers are exposed to a source corresponding to half the semi-infinite source.

$0.5 \cdot C_{Ra} \cdot 0.46 \cdot 10^{-9}$ Gy h^{-1} $\cdot$ 0.7 Sv Gy^{-1}
$\cdot$ 1 500 h a^{-1} = 10^{-3} Sv a^{-1}
=> C_{Ra} = 4 140 Bq kg^{-1}

The activity concentrations C_{Th}= 3070 Bq kg^{-1}, C_K = 46 500 Bq kg^{-1} and C_{Cs} = 12 700 Bq kg^{-1}

are calculated similarly, leading to the activity index I_4 for handling of peat ash presented in Section 2.3.

Use of peat ash in concrete

The amount of peat ash in concrete is 120 $kg \cdot m^{-3}$. The activity concentration of caesium in concrete C_{Cs} is

$$0.29 \cdot C_{Cs} \cdot 10^{-9}\ Gy\ h^{-1} \cdot 0.7\ Sv\ Gy^{-1} \cdot 7\,000\ h\ a^{-1} = 10^{-4}\ Sv\ a^{-1}$$
$$\Rightarrow C_{Cs} = 70\ Bq\ kg^{-1}$$

The activity concentration of caesium in peat ash is 70 $Bq\ kg^{-1} \cdot$ 2 300 kg/120 kg = 1 340 $Bq\ kg^{-1}$
$\Rightarrow$ 1000 $Bq\ kg^{-1}$

5.2 Indicators for elevated radioactivity of construction materials

The activity indexes presented above are used to assess whether the safety requirements are being fulfilled. For construction materials activity indexes are calculated on the basis of the measured activity concentrations of ^{226}Ra, ^{232}Th and ^{40}K. In special cases other nuclides are also taken into account, e.g. ^{137}Cs in fuel peat ash.

Measurements of activity concentrations are performed only if a reason exists for suspecting the presence of elevated radioactivity. It is not practical to measure all materials used. A problem is how to obtain the first hint of elevated radioactivity. One obvious criterion for performing measurements is if the raw material originates from an area known to have elevated levels of natural radioactivity.

Measurement of an external gamma dose rate by using a commonly available radiation survey meter can give some indication of the need for further investigations. It can be calculated that material whose activity index I_1 , I_2 or I_3 is equal to 1 will cause external gamma dose rates of 0.12 $\mu Gy\ h^{-1}$, 0.32 $\mu Gy\ h^{-1}$ or 0.92 $\mu Gy\ h^{-1}$, respectively, on an open field. In a least I_2 and I_3, some indication of the need for further investigation can be obtained with a direct dose rate measurement.

5.3 Assessments of occupational exposure

Handling, storage or any other operations involving materials containing elevated levels of natural nuclides always cause some excess exposure to the worker. The extent of exposure varies considerably in different operations depending on the exposure geometries, occupation times, dusting conditions etc. Exposure can not, therefore, be assessed solely on basis of the activity concentrations of the material.

For practical purposes it is useful to have an estimate for the levels of activity concentrations above which it is possible to exceed some predefined level of exposure. These levels of radioactivity in materials can be evaluated by assuming the presence of some extreme exposure conditions. Such conditions occur when the worker is continuously exposed to external gamma radiation from a semi-infinite source of that material and simultaneously the inhaled air contains dust originating from the material. The conditions for inhalation exposure can be described by the parameters given in Table *XIII*.

On these bases the activity concentrations which can, in extreme conditions, cause annual effective doses of 1 or 5 mSv are given in Table *XIV*. It can be concluded that in all normal working conditions the annual dose is less than 1 mSv if the ^{238}U and ^{232}Th concentrations, or their combinations, are below 1 000 $Bq\ kg^{-1}$.

Table XII. *General criteria and parameter values used in deriving activity indexes for construction materials and fuel peat ash.*

Conversion factor from absorbed dose in air to effective dose		0,7 Sv Gy^{-1}
Average dose rate outdoors due to natural radionuclides in undisturbed earth crust		70 nGy h^{-1}
	Indoors	**Outdoors**
Specific gamma dose rate	nGy h^{-1} per Bq kg^{-1}	nGy h^{-1} per Bq kg^{-1}
^{226}Ra	0.92	0.46
^{232}Th	1.1	0.62
^{40}K	0.080	0.041
^{137}Cs	0.29	0.15
Annual exposure/occupancy time	h	h
Dwellings (public)	7 000	
Workers	2 000	
Streets and playgrounds		500
Land filling areas		150

Table XIII. *Parameters used in assessing inhalation doses in a workplace.*

Inhalation rate	1.2	m^3 h^{-1}
Dust concentration in inhaled air	1	mg m^{-3}
AMAD of dust	1	µm
Nuclide or combination	**Dose conversion for inhalation**	
U++	55	µSv Bq^{-1}
Th++	68	µSv Bq^{-1}

Table XIV. *Activity concentrations of materials which can, in extreme conditions, cause an annual dose of 1 or 5 mSv to workers.*

Nuclide or combination	**1 mSv**	**5 mSv**
U++	1 300 Bq kg^{-1}	6 300 Bq kg^{-1}
Th++	1 000 Bq kg^{-1}	5 200 Bq kg^{-1}
K	17 000 Bq kg^{-1}	(83 000 Bq kg^{-1})
^{137}Cs	4 500 Bq kg^{-1}	23 000 Bq kg^{-1}

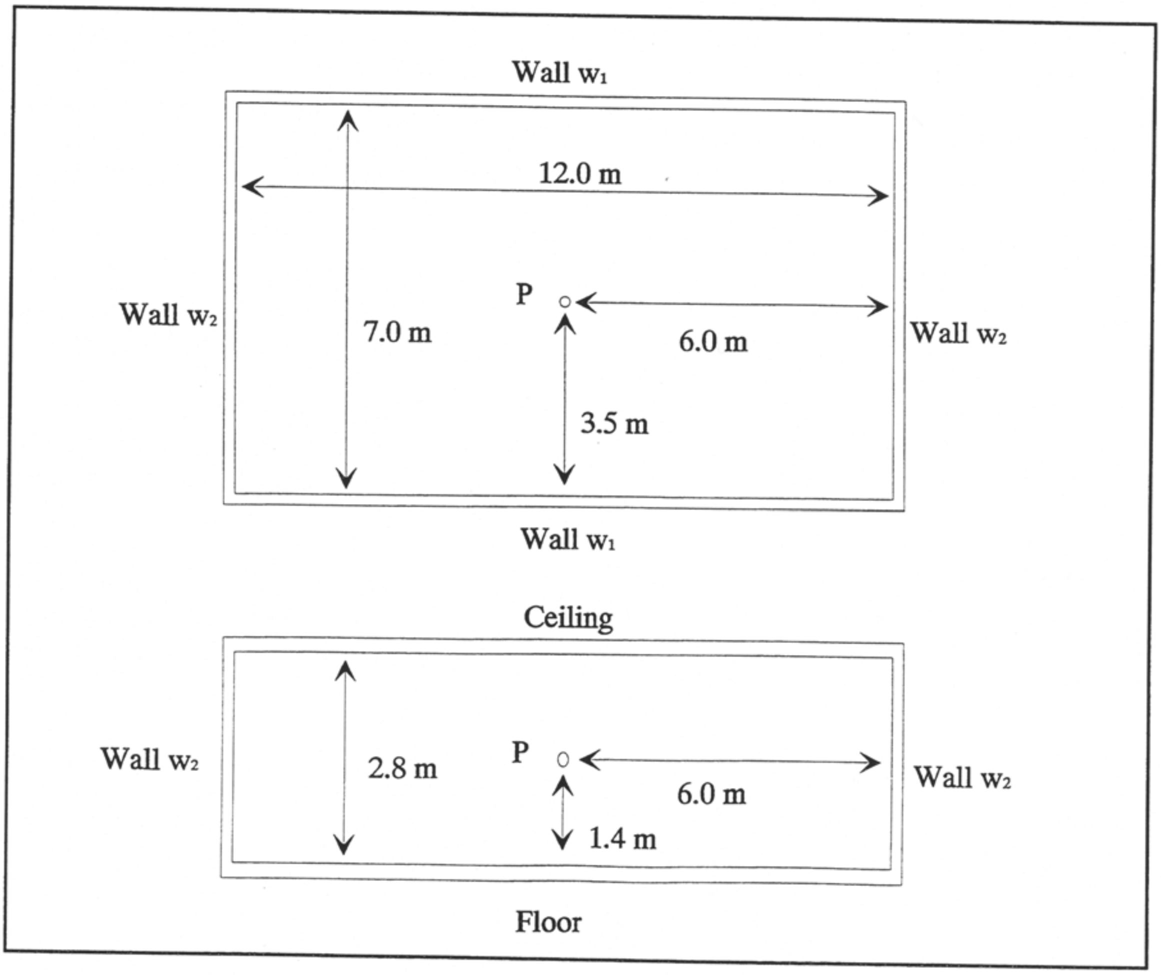

Figure 1. *The geometry used in calculation of indoor gamma dose rate from building materials.*

6 REFERENCES

1 ICRP Publication 60. 1990 Recommendations of the International Commission on Radiological Protection. Annals of the ICRP. Oxford: Pergamon Press, 1991.

2 Markkanen M. and Annanmäki M. Guidance on building materials and industrial waste containing elevated levels of radionuclides. Proceedings of International Symposium: Remediation and Restoration of Radioactive-contaminated Sites in Europe. Antwerp, 11-15 October 1993. Doc. XI-5074/94, Radiation Protection-74, European Commission.

3 Markkanen M. Safety guides for some practices causing public exposure to natural sources of radiation. Presented in International Seminar on Advancements in the implementation of the new basic safety standards-Experience in applying the 1990 recommendations of the ICRP. IAEA, Vienna, 20-24 November 1995.

4 Annanmäki M., Oksanen E. and Markkanen M. Radon at Workplaces Other than Mines and Underground Excavations. International Symposium on The Natural Radiation Environment (NRE VI), June 5-9. Montreal, Quebec, Canada.

5 Stranden E. Radioactivity of building materials and the gamma radiation in dwellings. Phys. Med. Biol. 1979; 24; 921-930.

6 Mustonen R. Methods for Evaluation of Radiation from Building Materials. Radiation Protection Dosimetry 1985; 7: 235-238.

7 Sources and Effects of Ionizing Radiation. United Nations Scientific Committee on the Effects of Atomic Radiation. UNSCEAR 1993 Report to the General Assembly, with Scientific Annexes. New York: United Nations, 1993.

8 Koblinger L. Calculation of exposure rate from gamma sources in walls of buildings. Health Physics 1978; 34:459-463.

9 Beck H.L, DeCampo J. and Gogolak G. In Situ Ge(Li) and NaI(Tl) Gamma-Ray Spectrometry. HASL-258 (1972).

10 Saito K. and Jacob P. Gamma ray fields in the air due to sources in the ground. Radiation Protection Dosimetry 1995; 58:29-45.

11 ICRP Publication 68. Dose Coefficients for Intakes of Radionuclides by Workers. Annals of the ICRP. Oxford: Pergamon Press, 1994.

12 ICRP Publication 65. Protection Against Radon-222 at Home and Work. Annals of the ICRP. Oxford: Pergamon Press, 1993.

13 Krisiuk E.M., et al. A study on radioactivity of building materials. Leningrad Research Institute for Radiation Hygiene, Leningrad, 1971.

14 Ingersoll, J.G. A survey of radionuclide contents and radon emanation rates in building materials used in the U.S. Health Physics 1983; 45:363-368.

15 Stranden E. and Berteig L. Radon in dwellings and influencing factors. Health Physics 1980; 39: 275-284.

16 Mustonen R. Natural radioactivity in and radon exhalation from Finnish building materials. Health Physics 1984; 46:1195-1203.

17 Markkanen M. Unpublished results of gammaspectrometric measurements performed by STUK as service for various producers of building materials in Finland during 1992 - 1995 (about 100 material samples).

18 Markkanen M. and Arvela H. Radon emanation from soils. Radiation Protection Dosimetry 1992;45:269-272.

19 Arvela H., Hyvönen H., Lemmelä H. and Castrén O. Indoor and outdoor gamma radiation in Finland. Radiation Protection Dosimetry 1995; 59, No 1: 25-32.

20 Arvela H., Markkanen M. and Lemmelä H. Mobile survey of environmental gamma radiation and fall-out levels in Finland after the Chernobyl accident. Radiation Protection Dosimetry 1990; 32:177-184.

21 Berger, M. J. Calculation of energy dissipation by gamma radiation near the interface between two media. Journal of Applied Physics 1957; 28: 1502-1508.

22 Trubey D.K. A survey of empirical functions used to fit gamma-ray build up factors, ORNL-RSIC-10, 1966.

23 Vrubel M., N. et. al. Build-up factors of scattered radiation from point source in an unbound air medium. Soviet atomic Energy 1973: 34; no1; 63-65.

Appendix 1

CALCULATION OF EXTERNAL GAMMA DOSE RATE

A1.1 Calculation of gamma dose rate

The geometry used in the calculation is shown in Figure A1. The absorbed dose rate in air D_1 ($Gy \cdot h^{-1}$) originating from the top layer at point P (x_p,y_p,z_p) can be calculated by using the formula:

$$D_1 = 5.77 \cdot 10^{-7} \frac{C_1 \rho_1}{4\pi} \sum \gamma_i \left(\frac{\mu_{en}}{\rho}\right)_i E_i \int B_i(1) \frac{e^{-\mu_i(1)s_1}}{l^2} dV$$

where the build-up factor $B_i(1)$ is written according to the Berger model[21-23]:

$$B_i(1) = 1 + C(E_i)\mu_i(1)s_1 e^{D(E_i)\mu_i(1)s_1}$$

and

$$s_1 = \left|\frac{z}{z_p - z}\right| l \qquad l = \sqrt{(x_p-x)^2+(y_p-y)^2+(z_p-z)^2}$$

and where the parameters are:

C_1	activity concentration of the top layer, Bq kg^{-1}
ρ_1	bulk density of the top layer, kg m^{-3}
γ_i	gamma intensity of gamma line i
E_i	gamma energy of gamma line i, MeV
$(\mu_{en}/\rho)_i$	energy absorption coefficient in air for gamma energy E_i , cm^2 g^{-1}
$\mu_i(1)$	attenuation coefficient in the top layer for gamma energy E_i , cm^{-1}
$C(E_i)$	coefficient in the Berger model, numerical values are given in Table *AI*
$D(E_i)$	coefficient in the Berger model, numerical values are given in Table *AI*
l	distance between point P (x_p,y_p,z_p) and the point of integration Q(x,y,z), cm
s_1	fraction of l within the top layer, cm
h_1	thickness of the top layer, cm
a, b	hight and width of the object, cm

The constant $5.77 \cdot 10^{-7}$ is derived from conversions of units ($3\,600$ s $h^{-1} \cdot 1.6 \cdot 10^{-13}$ MeV $J^{-1} \cdot 1\,000$ g $kg^{-1} = 5.77 \cdot 10^{-7}$).

The integration limits for the x-, y- and z-directions are -a/2...a/2, -b/2...b/2 and $-h_1$...0, respectively.

The absorbed dose rate in air D_2 at point P (x_p, y_p, z_p) due to the bottom layer can be calculated by using the formula

$$D_2 = 5.77 \cdot 10^{-7} \frac{C_2 \rho_2}{4\pi} \sum \gamma_i \left(\frac{\mu_{en}}{\rho}\right)_i E_i \int B_i(2) \frac{e^{-(\mu_i(1)s_1 + \mu_i(2)s_2)}}{l^2} dV$$

$$s_1 = \left|\frac{h_1}{z_p - z}\right| l \qquad s_2 = \left|\frac{z}{z_p - z}\right| l - s_1$$

where

s_2 is the fraction of l within the bottom layer, cm
$\mu_i(2)$ is the attenuation coefficient in the bottom layer for gamma energy E_i, cm^{-1}

The integration limits for the x-, y- and z-directions are -a/2...a/2, -b/2...b/2 and $-(h_1+h_2)$...0, respectively.

In this case, the estimate of the build-up factor is not so obvious as the attenuation term. The first approximation would be a product of the two build-up factors calculated separately for both layers. This would, however, overestimate the dose rate because the energy distribution of the flux has changed when the flux enters the upper layer. The following approximation is therefore used:

$$B_i(2) = B_i(1)\left(1 + \frac{\mu_i(2)s_2}{\mu_i(1)s_1 + \mu_i(2)s_2} \mu_i(2) s_2 e^{D(E_i)\mu_i(2)s_2}\right)$$

This is a product of the two build-up factors, but in the bottom layer only the fraction proportional to the mean free paths in the different layers is considered.

A1.2 Parameters for a simple computer program

In the cases of ^{238}U and ^{232}Th the absorbed dose rate should be calculated separately for every gamma line and then summed. It was demonstrated, however, by various comparisons that sufficient accuracy for practical assessments was achieved by using only one computational averaged gamma line for ^{238}U-series and two gamma lines for ^{232}Th-series. This is possible because the energy absorption and the attenuation coefficients are rather smooth functions of energy in the interval 240 - 1 800 keV. Only the intensive 2 615 keV gamma line of the thorium series is treated separately because this single line causes over 40% of the thorium series dose rate.

The average gamma energy was calculated by using the emission probability as a weighting factor. The emission probability for the computational gamma line is the sum of the emission probabilities. The values for energy absorption, attenuation and the energy-dependent coefficients in the build-up factor were chosen on the basis of the calculated average gamma energy. The values used are given in Table *A1-I*. The gamma lines used in calculating the averaged values are given in Table *A1-II*.

A1.3 Program list

A computer program for calculating the dose rates was written on the basis of the above. The Fortran program MATERIA uses an IMSL subroutine DMLIN for the numerical integration. The data file MATERIA.DAT consists of the source specification (dimensions, activity concentrations, densities) and the data file MATPARA.DAT consists of the general parameters given Table *A1-I*.

```
PROGRAM MATERIA

INTEGER MAXFCN,N,IER
REAL A(3),B(3),AERR,RERR,VAL,DMLIN
REAL
MYYL(2,5),MYY(5),MYYE(5),RHO(2),EN(5),DV
AK,VAK,GAMOS(5),C(2,5)
REAL
BCC(5),BDD(5),XX(4),ANN(2,5),OMANN(2,5)
REAL MYYLIN(2),XP,YP,ZP,H(2)
COMMON MYYLIN,H,BC,BD,XP,YP,ZP,RHO
EXTERNAL F,G

OPEN(3,FILE='MATERIA.DAT')
READ (3,*) (XX(I),I=1,4)
READ (3,*) XP,YP,ZP
READ (3,*) RHO(1),RHO(2)
READ (3,*) (C(1,I),I=1,4)
READ (3,*) (C(2,I),I=1,4)
CLOSE(3)

OPEN(3,FILE='MATPARA.DAT')
READ (3,*) (EN(I),I=1,5)
READ (3,*) (GAMOS(I),I=1,5)
READ (3,*) (MYY(I),I=1,5)
READ (3,*) (MYYE(I),I=1,5)
READ (3,*) (BCC(I),I=1,5)
READ (3,*) (BDD(I),I=1,5)
READ (3,*) DVAK
READ (3,*) MAXFCN,AERR,RERR
CLOSE(3)

C(1,5)=C(1,4)
C(1,4)=C(1,3)
C(1,3)=C(1,2)
C(2,5)=C(2,4)
C(2,4)=C(2,3)
C(2,3)=C(2,2)
H(1)=XX(3)
H(2)=XX(4)

DO 5 I=1,5
DO 7 J=1,2
MYYL(J,I)=RHO(J)*MYY(I)/2.35
7 CONTINUE
5 CONTINUE

RHO(1)=RHO(1)/1000
RHO(2)=RHO(2)/1000

C BOTTOM LAYER

A(1)=-XX(1)/2
A(2)=-XX(2)/2
A(3)=-(XX(3)+XX(4))
B(1)=XX(1)/2
B(2)=XX(2)/2
B(3)=-XX(3)

IF (H(2).EQ.0) GO TO 200

DO 10 I=1,5
```

```
VAK=DVAK*GAMOS(I)*C(2,I)*RHO(2)*MYYE
(I)*EN(I)/12.57
MYYLIN(1)=MYYL(1,I)
MYYLIN(2)=MYYL(2,I)
BC=BCC(I)
BD=BDD(I)
N = 3
IER=0
VAL=DMLIN(F,A,B,N,MAXFCN,AERR,RERR,IER
)
IF (IER.EQ.0) WRITE (6,*),'INTEGRATION OK'
IF (IER.EQ.129) WRITE (6,*),'INTEGRATION
NOT OK'
ANN(2,I)=1E6*VAK*VAL
OMANN(2,I)=1E9*VAL*VAK/C(2,I)
10  CONTINUE

C  TOP LAYER

200  A(3)=-H(1)
B(3)=0

DO 20 I=1,5
VAK=DVAK*GAMOS(I)*C(1,I)*RHO(1)*MYYE
(I)*EN(I)/12.57
MYYLIN(1)=MYYL(1,I)
BC=BCC(I)
BD=BDD(I)
N = 3
IER=0
VAL=DMLIN(G,A,B,N,MAXFCN,AERR,RERR,I
ER)
IF (IER.EQ.0) WRITE (6,*),'INTEGRATION OK'
IF (IER.EQ.129) WRITE (6,*),'INTEGRATION
NOT OK'
ANN(1,I)=1E6*VAK*VAL
OMANN(1,I)=1E9*VAL*VAK/C(1,I)
20  CONTINUE

IF (H(2).EQ.0) GO TO 300

WRITE (6,*)
WRITE (6,*),'TOTAL DOSE RATE, uGy/h '
TU1=ANN(1,1)+ANN(1,2)+ANN(1,3)+ANN(1,4)+
ANN(1,5)
TU2=ANN(2,1)+ANN(2,2)+ANN(2,3)+ANN(2,4)+
ANN(2,5)
TU=TU1+TU2
WRITE (6,*),TU
WRITE (6,*)
WRITE (6,*),'FRACTIONS, TOP AND
BOTTOM',TU1,TU2
WRITE (6,*)
WRITE (6,*),'FRACTIONS:'
WRITE (6,*)
WRITE (6,*),'URANIUM, TOP ja BOTTOM
LAYERS:',ANN(1,1),ANN(2,1)
TU1=ANN(1,2)+ANN(1,3)
TU2=ANN(2,2)+ANN(2,3)
WRITE (6,*),'THORIUM, TOP AND BOTTOM
LAYERS:',TU1,TU2
WRITE (6,*),'POTASSIUM, TOP AND
BOTTOM LAYERS:',ANN(1,4),ANN(2,4)
WRITE (6,*),'CESIUM, TOP AND BOTTOM
LAYERS:',ANN(1,5),ANN(2,5)
WRITE (6,*)
WRITE (6,*)
WRITE (6,*),'SPECIFIC DOSE RATES, nGy/h /
Bq/kg'
WRITE (6,*)
WRITE (6,*),'URANIUM, TOP AND BOTTOM
LAYERS:',OMANN(1,1),OMANN(2,1)
TU1=OMANN(1,2)+OMANN(1,3)
TU2=OMANN(2,2)+OMANN(2,3)
WRITE (6,*),'THORIUM, TOP ja BOTTOM
LAYERS:',TU1,TU2
WRITE (6,*),'POTASSIUM, TOP ja BOTTOM
LAYERS:',OMANN(1,4),OMANN(2,4)
WRITE (6,*),'CESIUM, TOP ja BOTTOM
LAYERS:',OMANN(1,5),OMANN(2,5)
GOTO 400

300   WRITE (6,*)
WRITE (6,*),'TOTAL DOSE RATE, uGy/h '
TU1=ANN(1,1)+ANN(1,2)+ANN(1,3)+ANN(1,4
)+ANN(1,5)
WRITE (6,*),TU1
WRITE (6,*)
WRITE (6,*)
WRITE (6,*),'FRACTIONS:'
WRITE (6,*)
WRITE (6,*),'URANIUM:',ANN(1,1)
TU1=ANN(1,2)+ANN(1,3)
WRITE (6,*),'THORIUM:',TU1
WRITE (6,*),'POTASSIUM:',ANN(1,4)
WRITE (6,*),'CESIUM:',ANN(1,5)
WRITE (6,*)
WRITE (6,*)
WRITE (6,*),'SPECIFIC DOSE RATES, nGy/h /
Bq/kg'
```

```
WRITE (6,*)
WRITE (6,*),'URANIUM:',OMANN(1,1)
TU1=OMANN(1,2)+OMANN(1,3)
WRITE (6,*),'THORIUM:',TU1
WRITE (6,*),'POTASSIUM:',OMANN(1,4)
WRITE (6,*),'CESIUM:',OMANN(1,5)
400     END

REAL FUNCTION F(N,X)
REAL X(N)
REAL L,S(2),B(2),R(2),SS
REAL
MYYLIN(2),H(2),BC,BD,XP,YP,ZP,RHO(2)
COMMON MYYLIN,H,BC,BD,XP,YP,ZP,RHO
L=SQRT((XP-X(1))*(XP-X(1))+(YP-X(2))*(YP-X
(2))+(ZP-X(3))*(ZP-X(3)))
SS=L*ABS(X(3)/(ZP-X(3)))
S(1)=L*ABS(H(1)/(ZP-X(3)))
S(2)=SS-S(1)
R(1)=MYYLIN(1)*S(1)
R(2)=MYYLIN(2)*S(2)
B(1)=BC*R(1)*EXP(BD*R(1))
B(2)=BC*R(2)*EXP(BD*R(2))
RR=R(1)+R(2)
BB=(1+B(1))*(1+R(2)*B(2)/RR)
F=BB*EXP(-R(1)-R(2))/(L*L)
RETURN
END

REAL FUNCTION G(N,X)
REAL X(N)
REAL L,S(2),B(2),R(2),SS
REAL
MYYLIN(2),H(2),BC,BD,XP,YP,ZP,RHO(2)
COMMON MYYLIN,H,BC,BD,XP,YP,ZP,RHO
L=SQRT((XP-X(1))*(XP-X(1))+(YP-X(2))*(YP-X
(2))+(ZP-X(3))*(ZP-X(3)))
SS=L*ABS(X(3)/(ZP-X(3)))
R(1)=MYYLIN(1)*SS
B(1)=1+BC*R(1)*EXP(BD*R(1))
G=B(1)*EXP(-R(1))/(L*L)
RETURN
END
```

<u>Data file MATERIA.DAT</u>

250	500	1.5	0
0	0	250	
0.5	0		
1	1	1	100
1	1	1	1

Explanations:
1. row: Source dimensions; width, hight, thickness(top, bottom), cm
2. row: Co-ordinates (x,y,z) of point P in which the dose rate is calculated (origin is in middle point on the surface of the top layer, z co-ordinate is the rectangular distance), cm
3. row: Densities of top and bottom layers, g cm^{-3}
4. row: Activity concentration; top layer: Uranium, Thorium, Potassium, Cesium
5. row: Activity concentration; bottom layer: Uranium, Thorium, Potassium, Cesium

Data file MATPARA.DAT

0.81	0.587	2.615	1.461	0.662
2.12	2.05	0.356	0.107	0.852
0.166	0.193	0.0927	0.124	0.183
0.0285	0.0295	0.0217	0.0257	0.0293
1.161	1.279	0.734	0.946	1.237
0.144	0.1900	0.0234	0.0755	0.1737
5.77E-7				
500 000	0	0.05		

Order of columns: Uranium, Thorium (low energies), Th (single line), Potassium and Cesium

1. Averaged gamma energies, MeV
2. Computational emission probability, dimensionless
3. Linear absorption coefficients (in concrete), cm^{-1}
4. Energy absorption coefficient (in air), 10^{-5} cm^2g^{-1}
5. Coefficient C from the Berger model, dimensionless
6. Coefficient D from the Berger model, dimensionless
7. Dose rate constant, dimensionless
8. Maximum number of iterations, desired absolute error, desired relative error

Table A1-I. *Averaged gamma energies, attenuation coefficients in concrete, energy absorption coefficients in air, emission probabilities and coefficients C and D in the build-up factors.*

Nuclide	**Averaged values used in calculations**					
	Energy keV	**γ**	***μ*** **cm^{-1}**	***μ_e/ρ*** **10^{-5} cm^2g^{-1}**	**C**	**D**
^{238}U	810	2.12	0.166	0.0285	1.161	0.144
^{232}Th	587	2.05	0.193	0.0295	1.279	0.190
^{232}Th	2 615	0.356	0.0927	0.0217	0.734	0.0234
^{40}K	1 461	0.107	0.124	0.0257	0.946	0.0755
^{137}Cs	662	0.852	0.183	0.0293	1.237	0.1737

Table A1-II. *List of gamma emissions.*

Nuclide Parent	Daughter	Energy keV	Emission %	Nuclide Parent	Daughter	Energy keV	Emission %
^{238}U	^{214}Pb	53.2	1.1	^{232}Th	^{228}Ac	99.5	1.3
		242.0	7.12			129.1	2.23
		295.2	18.2			209.3	3.81
		351.9	35.1			270.2	3.44
		786.0	1.04			327.6	3.1
	^{214}Bi	609.3	44.6			338.3	11.26
		768.4	4.76			409.8	1.95
		806.2	1.2			463.0	4.5
		934.1	3.07			772.1	1.45
		1 120.3	14.7			794.8	4.34
		1 155.2	1.9			835.6	1.53
		1 238.1	5.78			911.1	26.6
		1 281.0	1.6			964.4	5.05
		1 377.7	4.8			968.8	16.23
		1 401.5	1.4			1 588.3	3.26
		1 408.0	2.5			1 630.7	1.53
		1 509.2	2.08		^{224}Ra	241.0	4.04
		1 729.6	2.9		^{212}Bi	727.3	6.65
		1 764.5	15.1			785.5	1.07
		1 847.4	2.2			1 620.6	1.38
		2 118.5	1.17		^{212}Pb	238.6	43.5
		2 204.2	4.98			300.1	3.27
		2 447.8	1.55		^{208}Tl	277.4	2.3
						583.4	30.6
						860.4	4.55
						2 614.5	35.6

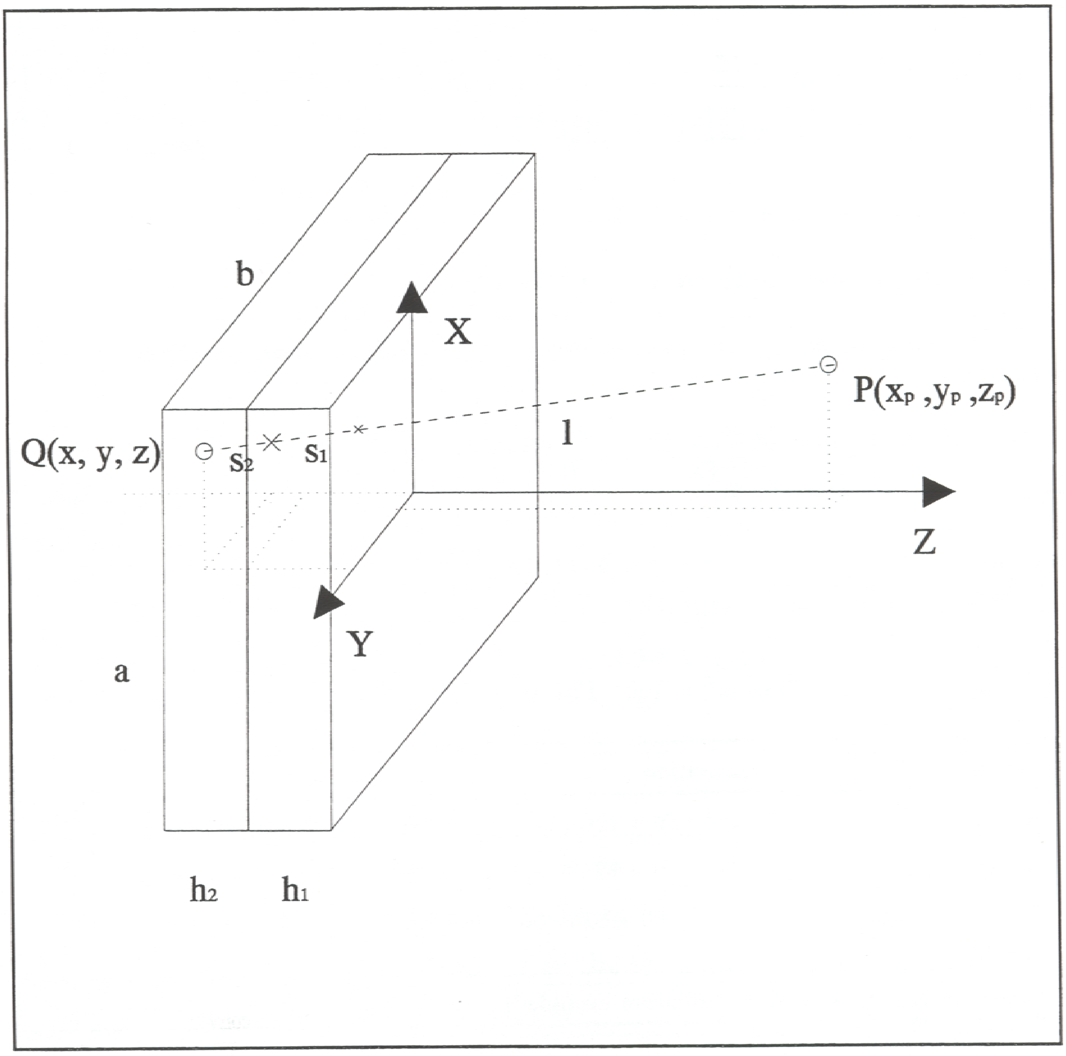

Figure A1. *The geometry used in calculation of external gamma dose rate.* Origin is the middle of the surface of the top layer.

Appendix 2

SAMPLE ASSESSMENTS USING SPECIFIC DOSE RATES IN TABLE FORM

A2.1 Gamma exposure in a concrete room

The walls, floor and ceiling of the room in Figure 1 (page 23) are of concrete having the following specifications:

Activity concentrations:	^{226}Ra	80 Bq kg^{-1}
	^{232}Th	80 Bq kg^{-1}
	^{40}K	800 Bq kg^{-1}
Density:		2 350 kg m^{-3}
Thickness:		20 cm

The specific mass of the walls is 2 350 kgm^{-3} · 0.20 m = 460 kgm^{-2}, thus the specific dose rates for a specific mass of 500 kg m^{-2} in Table *VIII* are used. The dose rate in the room is calculated:

Source	**Calculation**	**Dose rate**
W1, concrete	2 · (95 · 80 + 110 · 80 + 8.0 · 800)	0.0456 μGy h^{-1}
W2, concrete	2 · (32 · 80 + 37 · 80 + 2.7 · 800)	0.0154 μGy h^{-1}
Floor and ceiling, concrete	2 · (270 · 80 + 330 · 80 + 24 · 800)	0.1344 μGy h^{-1}
Dose rate in a room(cosmic radiation excluded)		**0.1954 μGy h^{-1}**

The annual dose to an occupant from the gamma radiation originating from the concrete is 0,7 Sv Gy^{-1} · 7 000 h · 0.1954 μGy h^{-1} = 957 μSv = 1 mSv. This is not, however, the excess exposure from building materials because concrete structures shield against gamma radiation from the undisturbed earth crust.

By using the average values for activity concentrations of the earth crust, 40 Bq kg^{-1} for ^{238}U and ^{232}Th and 700 Bq kg^{-1} for ^{40}K, and the specific dose rates given in Table *III*, we can calculate that the external gamma dose rate from the Earth's crust is (0.47 · 40 + 0.57 · 40 + 0.042 · 700) nGy h^{-1} = 0.071 μGy h^{-1}.

The excess gamma dose rate in the room is therefore (0.1954 - 0.071) μGy h^{-1} = 0.1244 μGy h^{-1} and the annual excess dose to the occupant is 0.7 Sv Gy^{-1} · 7 000 h · 0.1244 μGy h^{-1} = 610 μSv = 0.6 mSv. This is far below the safety requirement of 1 mSv and therefore, with regard to radioactivity, no restrictions on the use of such concrete are required.

A2.2 Gamma exposure in a concrete room with active tiles on the walls

The walls, floor and ceiling of the room in Figure 1 are of concrete and all the walls are covered with tiles containing elevated levels of natural radionuclides. The material specifications are:

	Concrete	**Tiles**
^{226}Ra	40 Bq kg^{-1}	200 Bq kg^{-1}
^{232}Th	40 Bq kg^{-1}	400 Bq kg^{-1}
^{40}K	700 Bq kg^{-1}	1 200 Bq kg^{-1}
Density	2 300 kg m^{-3}	2 500 kg m^{-3}
Thickness	20 cm	4 cm

The specific mass of the tiles is 2 500 kgm^{-3} · 0.04 m = 100 kgm^{-2}, so the corresponding values for specific dose rates in Table *VIII* are used.

The tiles on walls W1 (two similar walls) cause a dose rate:

Source nuclide	**Calculation**	**Dose rate**
^{226}Ra	2 · 35 pGy h^{-1} per Bq kg^{-1} · 200 Bq kg^{-1}	14 000 pGy h^{-1}
^{232}Th	2 · 40 pGy h^{-1} per Bq kg^{-1} · 400 Bq kg^{-1}	32 000 pGy h^{-1}
^{40}K	2 · 2.8 pGy h^{-1} per Bq kg^{-1} · 1 200 Bq kg^{-1}	6 700 pGy h^{-1}
Total dose rate caused by the tiles on walls W1		**0.0527 μGy h^{-1}**

Similarly it can be calculated for all sources in the room:

Source	Calculation	Dose rate
W1, tiles	2 · (35 · 200 + 40 · 400 + 2.8 · 1 200)	0.0527 μGy h^{-1}
W2, tiles	2 · (11 · 200 + 12 · 400 + 0.85 · 1 200)	0.0160 μGy h^{-1}
W1, concrete behind the tiles	2 · (67 · 40 + 77 · 40 + 5.6 · 800)	0.0205 μGy h^{-1}
W2, concrete behind the tiles	2 · (22 · 40 + 27 · 40 + 2.0 · 800)	0.0071 μGy h^{-1}
Floor and ceiling, concrete	2 · (270 · 40 + 330 · 40 + 24 · 800)	0.0864 μGy h^{-1}
Dose rate in the room (cosmic rad. excluded)		**0.1827 μGy h^{-1}**

By subtracting the gamma dose rate from the Earth's crust, 0.071 μGy h^{-1} (see Example 1), we can conclude that the excess gamma dose rate from all building materials in the room is (0.1827 -0.071) μGy h^{-1} = 0.1117 μGy h^{-1} and the annual excess dose to the inhabitant is 0,7 Sv Gy^{-1} · 7 000 h · 0.1117 μGy h^{-1} = 547 μSv = 0.5 mSv. This is far below the safety requirement of 1 mSv and therefore, with regard to radioactivity, no restrictions on the use of such materials are required.

In this example it is interesting to further analyze the origin of the gamma dose and especially the amount of the excess dose caused by the tiles. The dose rate caused by the tiles is 0.0527 μGy h^{-1} + 0.0160 μGy h^{-1} = 0.0687 μGy h^{-1}, however the excess caused by tiles is less than 0.0687 μGy h^{-1} because they reduce the dose rate caused by the concrete behind them. The excess caused by the tiles can, therefore, be obtained by calculating the dose rate in a room without tiles and by subtracting this from the total dose rate calculated above. The dose rate for the room without tiles is calculated as prescribed in Example 1, giving a value of 0.1254 μGy h^{-1}.

The excess dose rate caused by the tiles is (0.1827- 0.1254) μGy h^{-1} = 0.0573 μGy h^{-1} and the excess annual dose caused by the tiles to an occupant is 0.7 Sv Gy^{-1} · 7 000 h · 0.0573 μGy h^{-1} = 280 μSv = 0.3 mSv.

A2.3 Radon exposure caused by building materials

Let us consider the situation described in Example 2 and make the following assumptions. The tiles on the walls prevent the release of radon from the concrete structures behind them. Radon is released from the floor and ceiling structures. Let us assume further that all free radon produced in the tiles enters the room. The emanation coefficient of the tiles is 0.2 and the nominal exhalation rate from the floor and ceiling concrete is 0.4 Bq h^{-1} per Bq kg^{-1}. The air exchange rate of the room is 0.5 h^{-1}.

On the basis of the dimensions given in Example 2 we can be calculate that the volume of the room is 235 m^3, the combined area of the floor and ceiling is 168 m^2 and the total mass of tiles in the room is 9800 kg.

Floor and ceiling

By using Equation 3 we can calculate that the exhalation rate from the concrete floor and ceiling is 0.4 Bq $h^{-1}m^{-2}$ per Bq kg^{-1} · 40 Bq kg^{-1} · 168 m^2 = 2700 Bq h^{-1}, which causes an excess radon concentration of 2700 Bq h^{-1} /(0.5 h^{-1} · 235 m^3) = 23 Bq m^{-3}.

Tiles

By using the latter version of Equation 3 we can be calculate that the exhalation rate from the tiles is 0.00756 h^{-1} · 0.2 · 200 Bq kg^{-1} · 9800 kg = 3000 Bq h^{-1}, which causes an excess radon concentration 3000 Bq h^{-1} /(0.5 h^{-1} · 235 m^3) = 25 Bq m^{-3}.

By using the dose conversions given in Section 3.3 we can calculate that the excess annual effective dose caused by the concrete structures is 0.39 mSv in the home and 0.14 mSv at work. The tiles cause an excess dose of 0.43 mSv in the home and 0.16 mSv at work.

A2.4 Gamma exposure on a landfill site

Mine tailings are used for landfill purposes and are covered with a 20-cm-thick layer of soil. The annual occupancy time at the area is estimated to be 150 h. The material specifications are:

	Covering soil	**Tailing**
^{226}Ra	40 Bq kg^{-1}	200 Bq kg^{-1}
^{232}Th	40 Bq kg^{-1}	400 Bq kg^{-1}
^{40}K	700 Bq kg^{-1}	1 200 Bq kg^{-1}
Density	1 500 kg m^{-3}	1 800 kg m^{-3}
Thickness of layer	20 cm	> 50 cm

The specific mass of the covering soil is 1 500 kgm^{-3} · 0.2 m = 300 kgm^{-2}, thus the corresponding values for specific dose rates in Table *X* are used.

The covering soil causes a dose rate:

Source nuclide	Calculation	Dose rate
^{226}Ra	410 pGy h^{-1} per Bq kg^{-1} · 40 Bq kg^{-1}	16 400 pGy h^{-1}
^{232}Th	480 pGy h^{-1} per Bq kg^{-1} · 40 Bq kg^{-1}	19 200 pGy h^{-1}
^{40}K	35 pGy h^{-1} per Bq kg^{-1} · 700 Bq kg^{-1}	24 500 pGy h^{-1}
Total dose rate caused by the covering soil		**0.0601 μGy h^{-1}**

The tailings under the soil layer causes a dose rate:

Source nuclide	Calculation	Dose rate
^{226}Ra	62 pGy h^{-1} per Bq kg^{-1} · 200 Bq kg^{-1}	12 400 pGy h^{-1}
^{232}Th	91 pGy h^{-1} per Bq kg^{-1} · 400 Bq kg^{-1}	36 400 pGy h^{-1}
^{40}K	7.3 pGy h^{-1} per Bq kg^{-1} · 1200 Bq kg^{-1}	8 800 pGy h^{-1}
Total dose rate caused by the tailing material		**0.0576 μGy h^{-1}**

The total dose rate outdoors is 0.060 μGy h^{-1} + 0.058 μGy h^{-1} = 0.118 μGy h^{-1}. The average dose rate from the undisturbed Earth's crust is 0.071 μGy h^{-1} (see Example 1). The excess dose rate caused by the tailings is 0.118 μGy h^{-1} - 0.071 μGy h^{-1} = 0.047 μGy h^{-1}. The excess annual dose caused by the tailings to a person remaining in the area for 150 h is 0.7 Sv Gy^{-1} · 150 h · 0.047 μGy h^{-1} = 4.9 μSv. This is far below the safety requirement of 0.1 mSv.

A2.5 Gamma exposure caused by a pile of material in the a working area

Zircon sand containing 3 000 Bq kg^{-1} ^{238}U and 700 Bq kg^{-1} ^{232}Th is stored in sacks next to a working room. The sacks are piled so that the area facing towards the working room is about 4 m^2. The wall between the rooms is made of thin gypsum boards so absorbtion in it is not accounted for. The distance from the pile to the workplace averages 5 meters and the annual working time is 1 500 h.

The data of Table XI give the absorbed dose rate in air from the pile as 12 pGy h^{-1} per Bq kg^{-1} · 3 000 Bq kg^{-1} (^{238}U)+ 14 pGy h^{-1} per Bq kg^{-1} · 700 Bq kg^{-1} (^{232}Th) = 46 000 pGy h^{-1} = 0.046 μGy h^{-1}. The annual effective dose to the worker from the pile of material is 0.7 Sv Gy^{-1} · 0.046 μGy h^{-1} · 1 500 h = 48 μSv = 0.05 mSv.